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Summary Technical Report for Conceptual Design of MARCH-SERTTA

Nuclear Technology Research and Development

Prepared for
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SUMMARY

The Transient Reactor Test facility (TREAT) was constructed in the late 1950's, provided thousands of transient irradiations before being placed in standby in 1994, and resumed reactor operations in 2017 in order to reclaim its crucial role in nuclear-heated safety research [1]. Starting in September of 2018, TREAT performed its first series of new fueled transient experiments using a novel modular irradiation vehicle, termed the Minimal Activation Retrievable Capsule Holder (MARCH) system. The MARCH system was developed to support high throughput testing and post transient examination by using fresh fuel samples arranged in low-activation hardware materials. Deployment of this irradiation vehicle system was supported by the Accident Tolerant Fuels (ATF) program. These initial experiments used inert gas atmosphere capsules (referred to as the Separate Effects Test Holder, SETH) containing small rodlets composed of UO_2 pellets in zirconium alloy cladding with temperature measurement instruments. The initial SETH campaign was conducted in order to commission irradiation capabilities and deepen the understanding of specimen to core energy coupling. Building upon the successes of the initial SETH campaign, an evolution of the capsule design has been conceptualized in order to support the same kind of high throughput, cost effective, rodlet scale testing, but with the notable improvement of being able to immerse specimens in liquid water for testing conditions better representing Light Water Reactors (LWR). This capsule is referred to as the MARCH system Static Environment Rodlet Transient Test Apparatus (MARCH-SERTTA).

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1. INTRODUCTION

The Transient Reactor Test facility (TREAT) was constructed in the late 1950's, provided thousands of transient irradiations before being placed in standby in 1994, and resumed reactor operations in 2017 in order to reclaim its crucial role in nuclear-heated safety research [1]. Starting in September of 2018, TREAT performed its first series of new fueled transient experiments using a novel modular irradiation vehicle, termed the Minimal Activation Retrievable Capsule Holder (MARCH) system. The MARCH system was developed to support high throughput testing and post transient examination by using fresh fuel samples arranged in low-activation hardware materials. Deployment of this irradiation vehicle system was supported by the Accident Tolerant Fuels (ATF) program. These initial experiments used inert gas atmosphere capsules (referred to as the Separate Effects Test Holder, SETH) containing small rodlets composed of UO_2 pellets in zirconium alloy cladding with temperature measurement instruments. The initial SETH campaign was conducted in order to commission irradiation capabilities and deepen the understanding of specimen to core energy coupling. Building upon the successes of the initial SETH campaign, an evolution of the capsule design has been conceptualized in order to support the same kind of high throughput, cost effective, rodlet scale testing, but with the notable improvement of being able to immerse specimens in liquid water for testing conditions better representing Light Water Reactors (LWR). This capsule is referred to as the MARCH system Static Environment Rodlet Transient Test Apparatus (MARCH-SERTTA). An overview image of TREAT and the MARCH system are shown in Figure 1 and Figure 2, respectively.

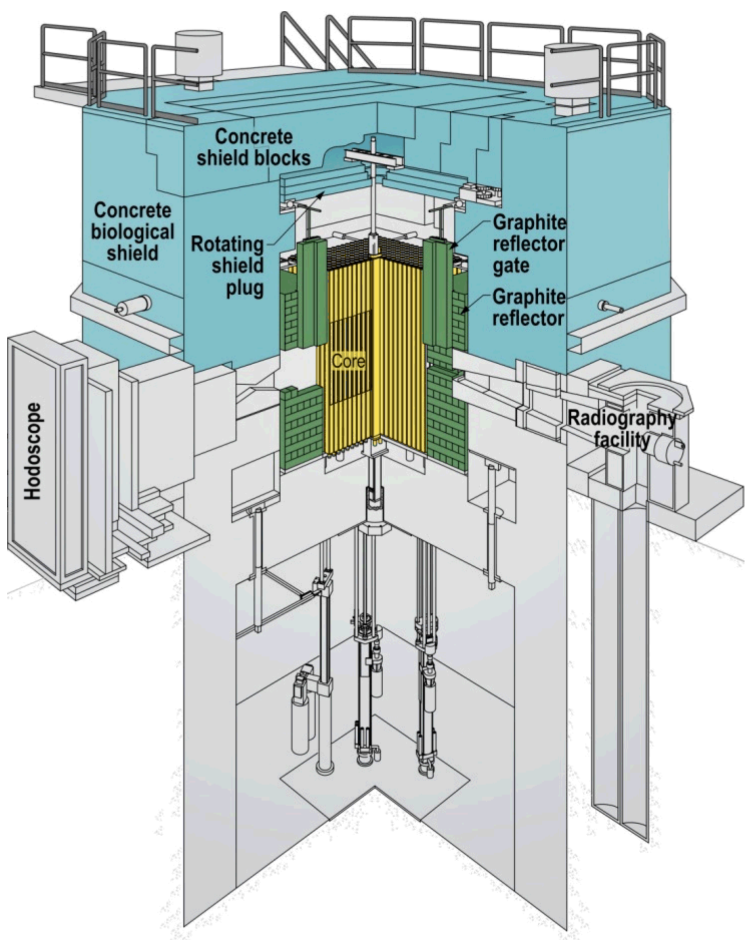


Figure 1. Overview of TREAT Features, ¾ Section View. [1]

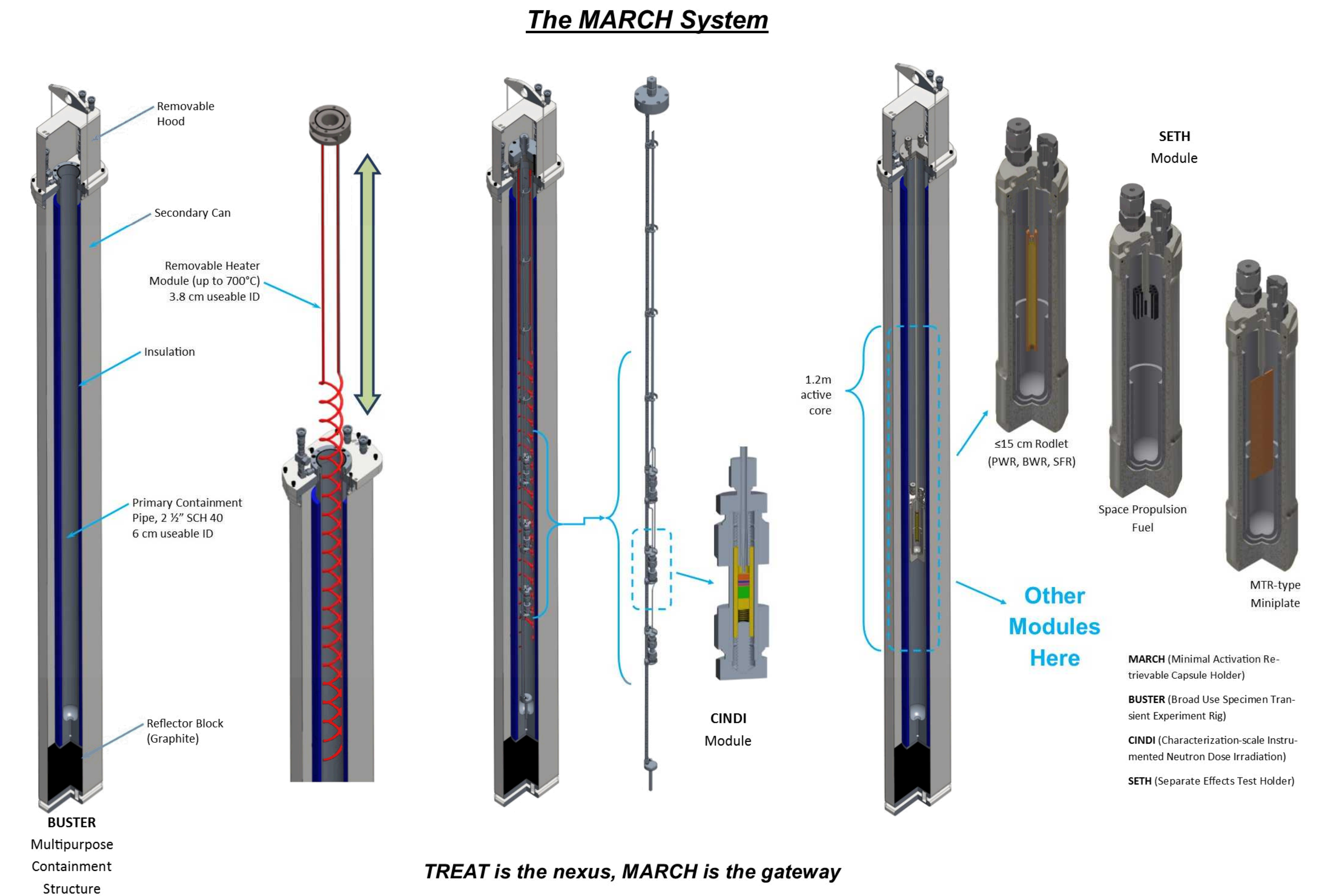


Figure 2. Overview of the MARCH System. [1]

The purpose of the MARCH-SERTTA device is to perform integral tests evaluating the Pellet Clad Mechanical Interaction (PCMI) performance of LWR fuel concepts in design basis Reactivity Initiated Accidents (RIA). In RIAs the prompt reactivity insertion causes a sharp increase in power that is terminated by the negative reactivity produced by Doppler feedback in the UO_2 pellets. The power pulse has a Gaussian shape characterized by a Full Width at Half Maximum (FWHM) and a decay power tail. The energy deposited in the pellet increases the pellet's enthalpy which is equal to the total energy deposited minus energy lost from heat transfer. Typical power, energy, and enthalpy curves are illustrated below in Figure 3. For LWRs the most limiting RIA transients are Control Rod Drop (CRD BWR) and Control Rod Ejections (CRE PWR) at zero power conditions in either cold or hot coolant. These transients are the most limiting in that they have the highest energy releases and the narrowest pulse widths.

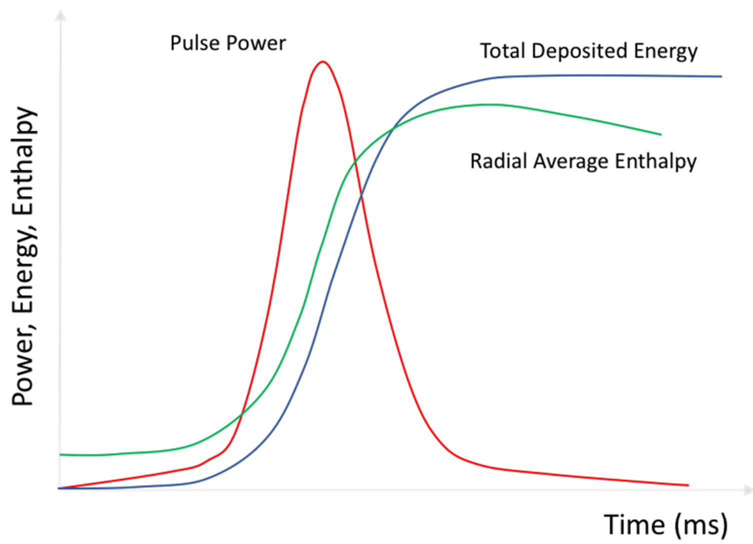


Figure 3. Typical Power, Energy, and Enthalpy curves in an LWR RIA.

In RIA events the rapid thermal expansion of the pellet can overstress the cladding causing rupture. PCMI failures are the result of a rapid displacement controlled bi-axial loading on the cladding. These conditions are difficult to simulate in out of pile experiments. PCMI failures are often termed early-phase failures as they take place early in the transient. As thermal expansion of the pellet is the cause of the rupture, the failure thresholds are often expressed as a maximum radial average enthalpy increase at the peak axial fuel location. For PCMI failures only the prompt energy release needs to be considered in calculating the pellet enthalpy. The prompt energy release includes the energy released up to one Full Width and Half Max (FWHM) after the peak power [2]. Failures after this time are the result of elevated cladding temperature causing balloon and burst type ruptures, as well as cladding embrittlement due to high temperature oxidation in steam following a departure from nucleate boiling.

PCMI failure in an RIA for traditional UO2-Zircaloy fuel rods is a burnup enhanced failure mode. As burnup in LWR fuel increases the cladding is embrittled due to irradiation damage and hydrogen uptake. In addition, the pellet cladding gap closes due to pellet swelling and cladding creep down. Of these, excess cladding hydrogen is the first order burnup related effect which increases the fuel rod's propensity to experience PCMI failure. The effect is more severe for claddings with recrystallized annealed heat treatments such as Zr-2. Hydrides in these claddings have a random orientation as opposed to the dominantly circumferential orientation found in stress relieved annealed cladding such as Zr-4 and Zirlo. The test results also show an increased propensity for PCMI failures as low coolant temperatures. The NRC's draft regulatory guide expresses these dependencies as a family of four curves illustrated in below Figure 4 [3].

Figure 4. PCMI Failure Limits for Zircaloy Cladding in RIA Transients.

The displacement-controlled loadings may be even more important for emerging Accident Tolerant Fuel (ATF) concepts such as advanced steel (FeCrAl) and silicon carbide composite (SiC-SiC) claddings. While these cladding likely exhibit an increased resistance to high temperature oxidation environments experienced in the later phase of RIAs and in LOCAs, their mechanical properties may have a lower ability to accommodate applied displacements [4]. Testing in MARCH-SERTTA for these ATF claddings as well as high burnup Zircaloy claddings is crucial to establishing PCMI performance limits in RIAs.

2. MECHANICAL DESIGN

The MARCH-SERTTA design builds upon the SETH capsule essentially by lengthening the capsule region to the maximum size possible for most available Direct Metal Laser Sintering (DMLS) build chambers. Use of additive manufacturing enables the mostly-hollow geometry to be produced more economically and without the difficulty often encountered with welding. A titanium alloy compositionally equivalent to conventional grade 5 (UNS R56400) is used for its respectable strength, well established DMLS parameters already developed from its use in aerospace and medical industries, and lack of constituents which transmute into highly radioactive isotopes during neutron irradiation. Like the SETH design, MARCH-SERTTA uses hemisphere ended mullite crucibles to protect the capsule wall from molten fuel thermal attack. A leftover mullite crucibles from the SETH campaign was tested in pressurized hot water autoclave to verify its survival in these conditions.

Unlike SETH, the MARCH-SERTTA lid approximately the same size as the capsule itself. This geometry is also designed to be manufactured by DLMS to give a large internal gas cavity. This expansion volume is the principle modification which enables water-environment testing. While the capsule lid is designed to accommodate pressure increases during transient testing, an integral burst disc on the top of the lid enables MARCH-SERTTA to relieve pressurize into the much larger Broad Use Specimen Transient Experiment Rig (BUSTER) pipe in the event of unexpected pressure rise. A small commercially-available pressure transducer is placed on top of the lid to monitor internal capsule pressure. Three compression seal fittings for instrument lead penetrations are also placed on top of the capsule lid; enabling up to twelve 1mm instrument leads to penetrates into the capsule. A hanger rod interface similar to that used for

SETH connects to a closure flange which is compatible with the existing BUSTER pipe. See Figure 5.

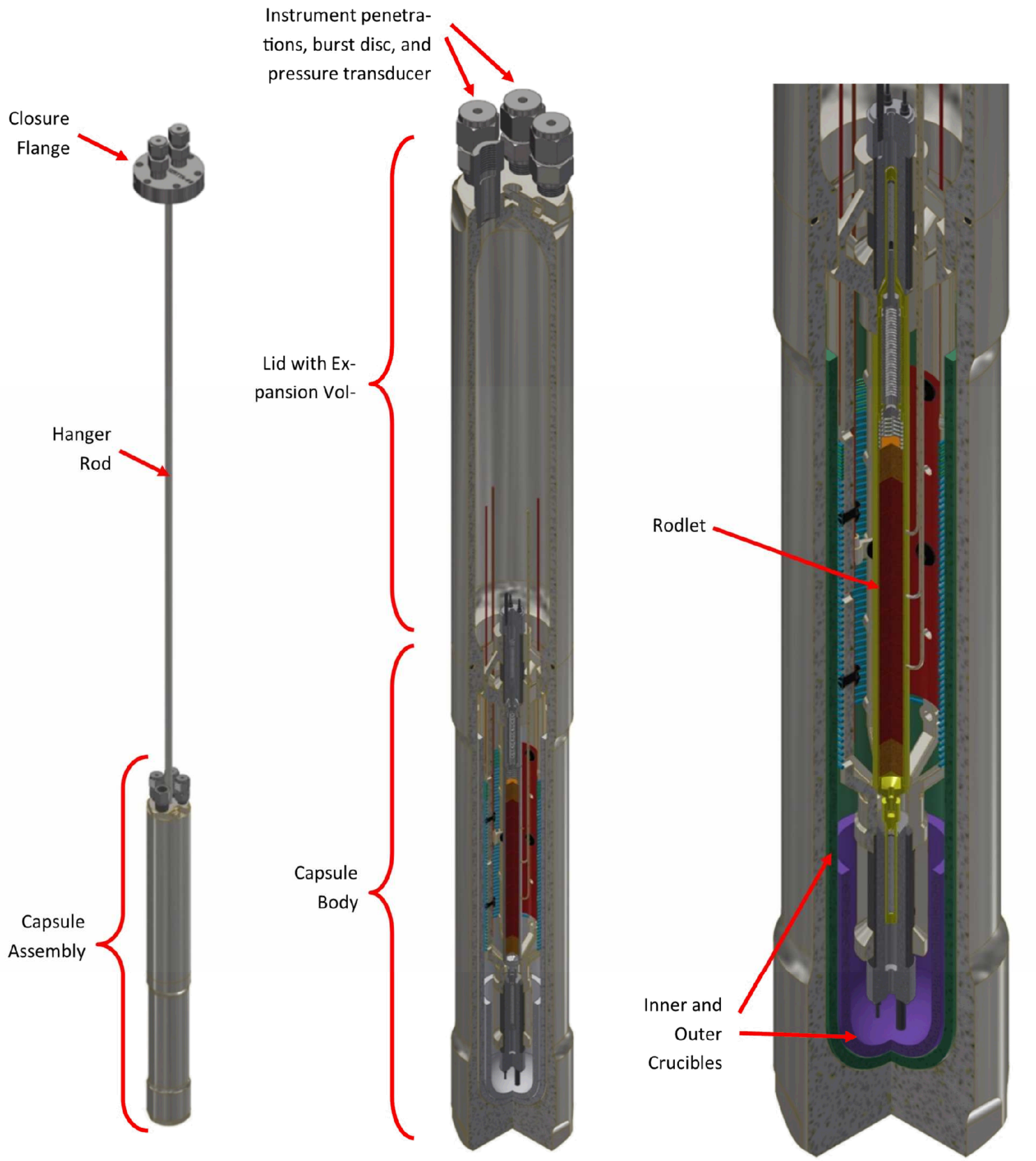


Figure 5. MARCH-SERTTA Module Overview.

The MARCH-SERTTA rodlet and instrumentation package are supported from a skeleton-frame holder suspended from the bottom of the lid with small screws. This holder is also designed to be manufactured by DMLS, but its material of construction is still under investigation to determine the optimum characteristics for specimens and instrument interaction. This holder can support several instruments and be easily configured for different testing purposes. The present design concept is arranged to support RIA testing of a ten pellet rodlet within metallic cladding at typical pressurized water reactor diameters. The eight central pellets are at the full design ^{235}U enrichment, while the top and bottom pellets are either fuel pellets of a lower enrichment, or non-fueled ceramic insulator pellets, in order to help reduce anomalous end effects in fuel performance. The rodlet design is based on an established approach where an internal bellows in the upper plenum deforms as pressure increases thus moving a ferritic wire downward so that Linear Variable Differential Transformer (LVDT) instrumentation can detect pressure change. Similarly, an encapsulated ferritic wire is attached to the bottom cladding end cap so that and LVDT can detect cladding elongation. The lower end cap of the rodlet is designed so that inert gas can be pressurized through small hole into the rodlet and seal welded during fabrication. The rodlet design is shown in Figure 6.

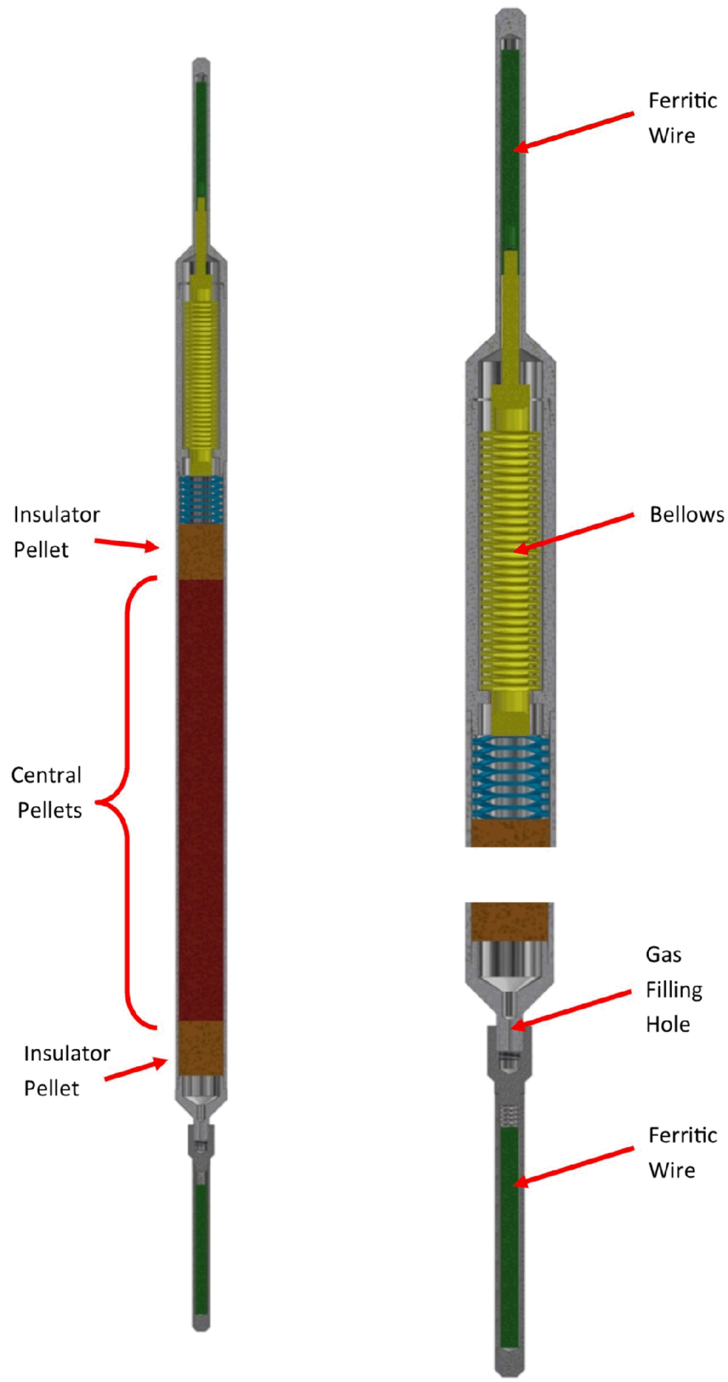


Figure 6. MARCH-SERTTA RIA Rodlet Design.

In addition to supporting the rodlet and two LVDTs described previously, the instrumentation and specimen holder also supports several other instruments. A few thermocouples (TC) are attached to cladding surfaces by spot welding. An optical fiber also views the cladding surface for fast-response non-contact pyrometry. Both of these approaches were successfully employed in the SETH capsule for cladding temperature measurement, although lab testing and optimization of TC type/welding parameters will be needed to fully access high confidence data in water environment testing. The same is true of material selection and data acquisition/interpretations parameters for optical measurements. Two semi-cylindrical metal plates are also fastened to the frame on either side of the rodlet so that change in capacitance between them can be correlated to water density for boiling detection. These plates are coated in a high temperature polymer to minimize noise that would otherwise arise from conductive contact with water. This approach was tested successfully in out-of-reactor prototypes during other previous design efforts. Lastly, a small diameter electric cable heater and multipoint TC is wrapped around the instrument holder frame to enable water temperature to be elevated prior to transient initiation. A cross section of the specimen/instrument array and an overview rendering of the assembly can be seen in Figure 7 and Figure 8, respectively.

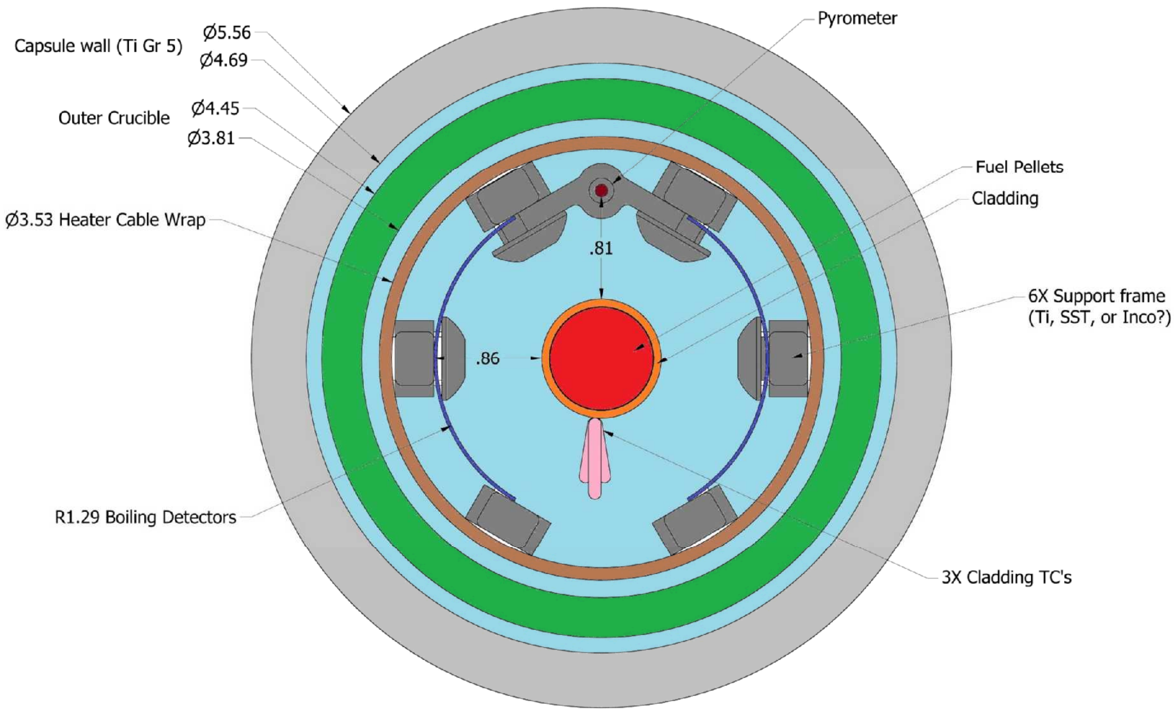


Figure 7. Instrument and Specimen Cross Section.

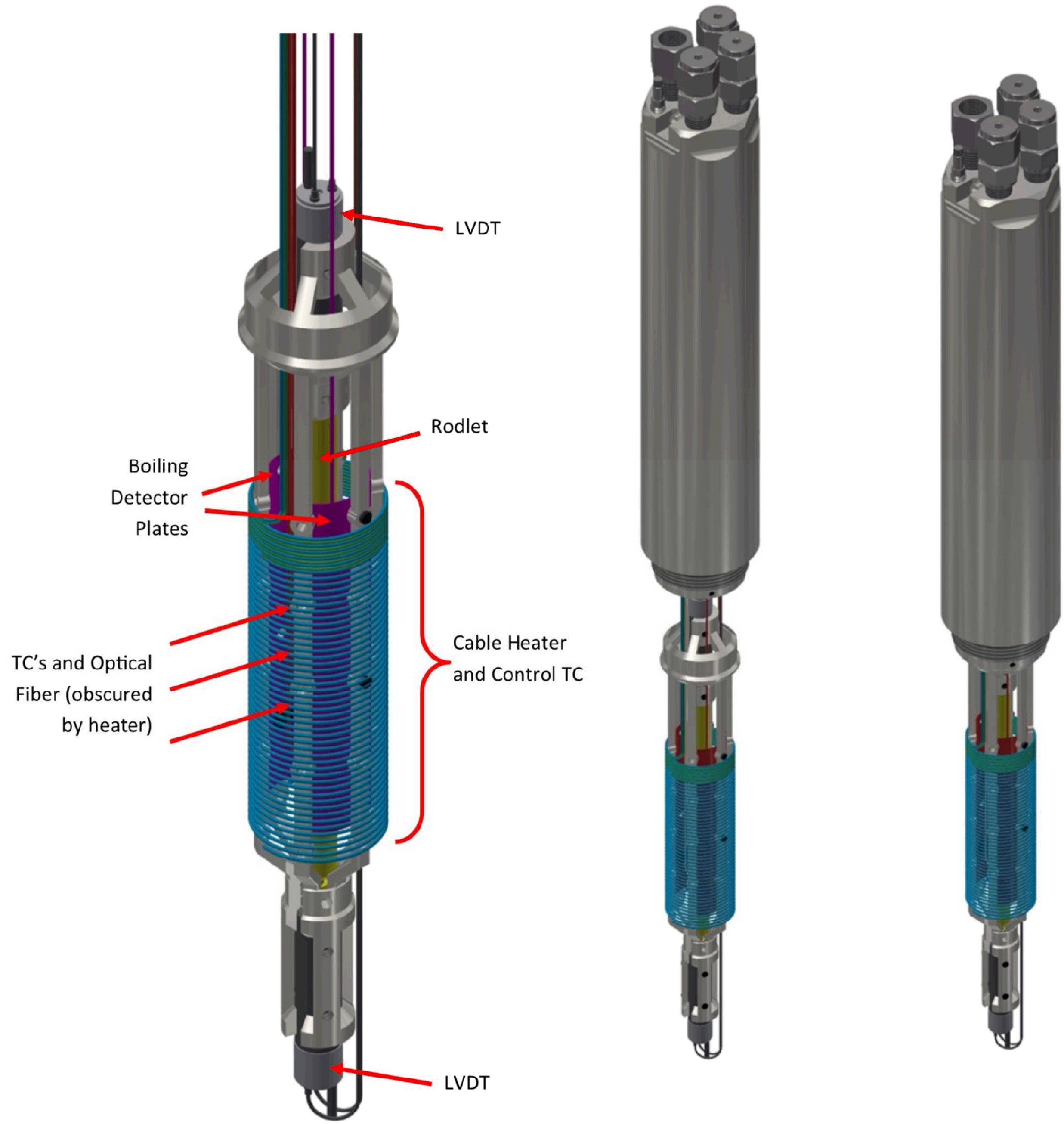


Figure 8. Instrument and Specimen Holder Overview.

3. NEUTRONIC CALCULATIONS

Neutronics analysis of the MARCH-SERTTA design was performed using Monte Carlo n-Particle (MCNP) code, version 6.1 [5] with ENDF/BVII.1 nuclear data cross sections [6]. As ENDF/BVII.1 only has a few selections of temperatures at which cross sections have been evaluated, custom cross sections for temperatures ranging from 20°C to 320°C were created using the *makxsf* utility included with MCNP. The process found in ECAR-3994 [7] was used to generate the custom cross sections needed for analysis of the design. The model used in the validation and verification of MCNP6.1 for TREAT found in ECAR-3846 [8] was chosen as the basis for the TREAT reactor model. This model was selected as it is based on the SCALE neutronics code model used by TREAT Reactor Engineering. The materials in the MCNP model were updated in this work from those used in the ECAR-3846 version as several errors in the material specifications had been identified. Additional materials for MARCH-SERTTA were added as well. A cutoff window approximately enclosing the permanent reflector and reactor core was also emplaced to better match modeled TREAT extents used in previous work done on the MARCH-SETH design analysis [7]. The core was modeled in a full slot configuration, see Figure 9, which will allow for better use of the hodoscope and to limit the excess reactivity in the core during actual transient testing.

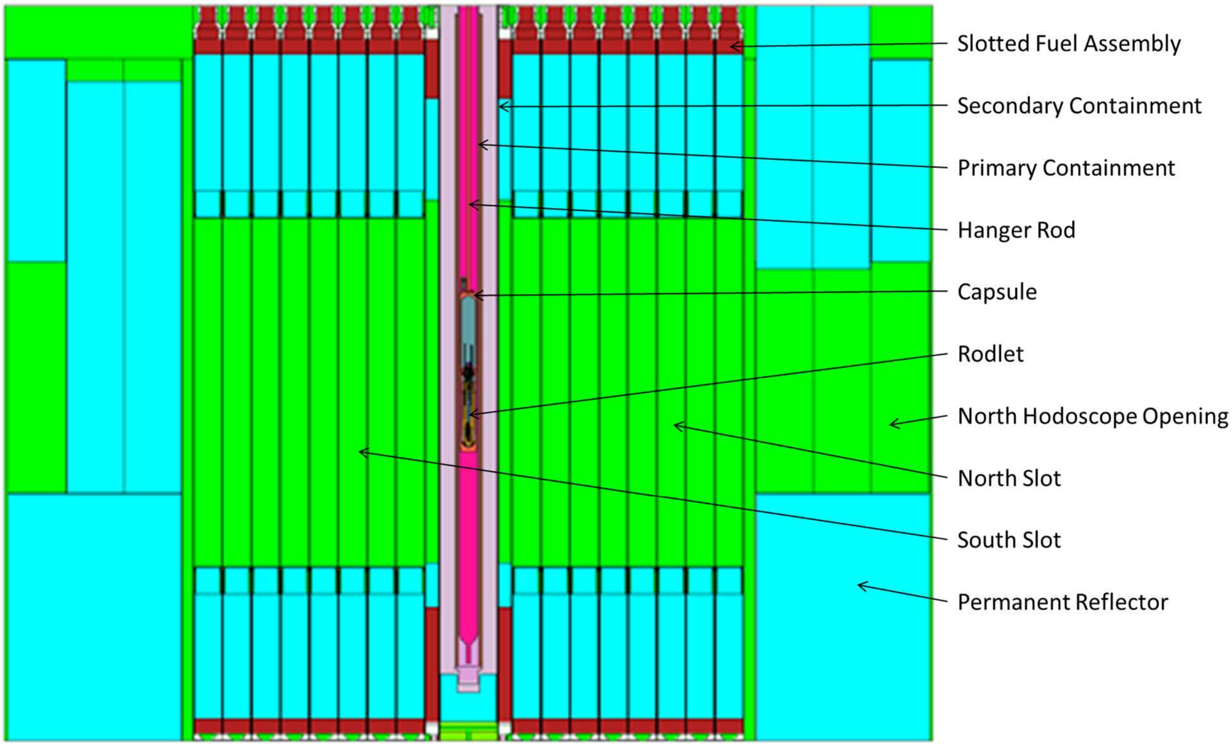


Figure 9. Vertical cross section (South-North) of TREAT reactor core showing MARCH-SERTTA positioning.

The test section of the model consisted of the secondary and primary containment structure of the BUSTER as well as the MARCH-SERTTA capsule. For this analysis, the secondary and primary containments were considered part of the reactor while the primary containment cover gas, the capsule, and everything inside the capsule were considered to be the experiment regarding the assignment of component temperatures. The containments and the experiment were modeled as closely as possible to the actual design with minimal simplification in an attempt to capture any spatial effects; the capsule being shown in Figure 10. Because of the complexity of some parts, simplification was necessary to work within the limitations of MCNP. These simplifications were made in such a way to best preserve the volume (thus conserving the mass) of the parts and spatial configuration if possible.

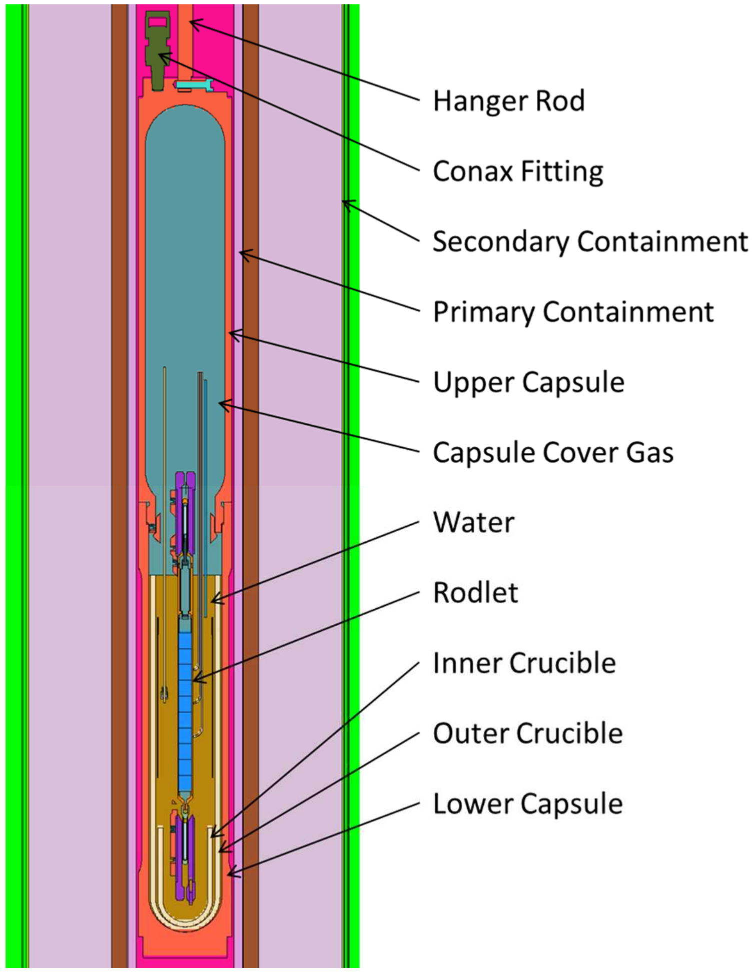


Figure 10. Vertical cross section (South-North) of MARCH-SERTTA capsule.

The model of the rodlet was a 10 pellet stack of fresh UO_2 in Zircaloy cladding with a helium cover gas. The pellets were modeled as having a diameter of 0.8255 cm and a height of 1.016 cm to approximate PWR fuel pellets. The maximum theoretical density of 10.97 g/cm^3 for UO_2 was used as a conservative value in calculating UO_2 composition. Heat generation rates (HGRs) and power coupling factors (PCFs) were calculated using an F6 type tally in MCNP, predicting energy deposition in the materials. Figure 11 and Figure 12 provide detailed views of the rodlet within the capsule.

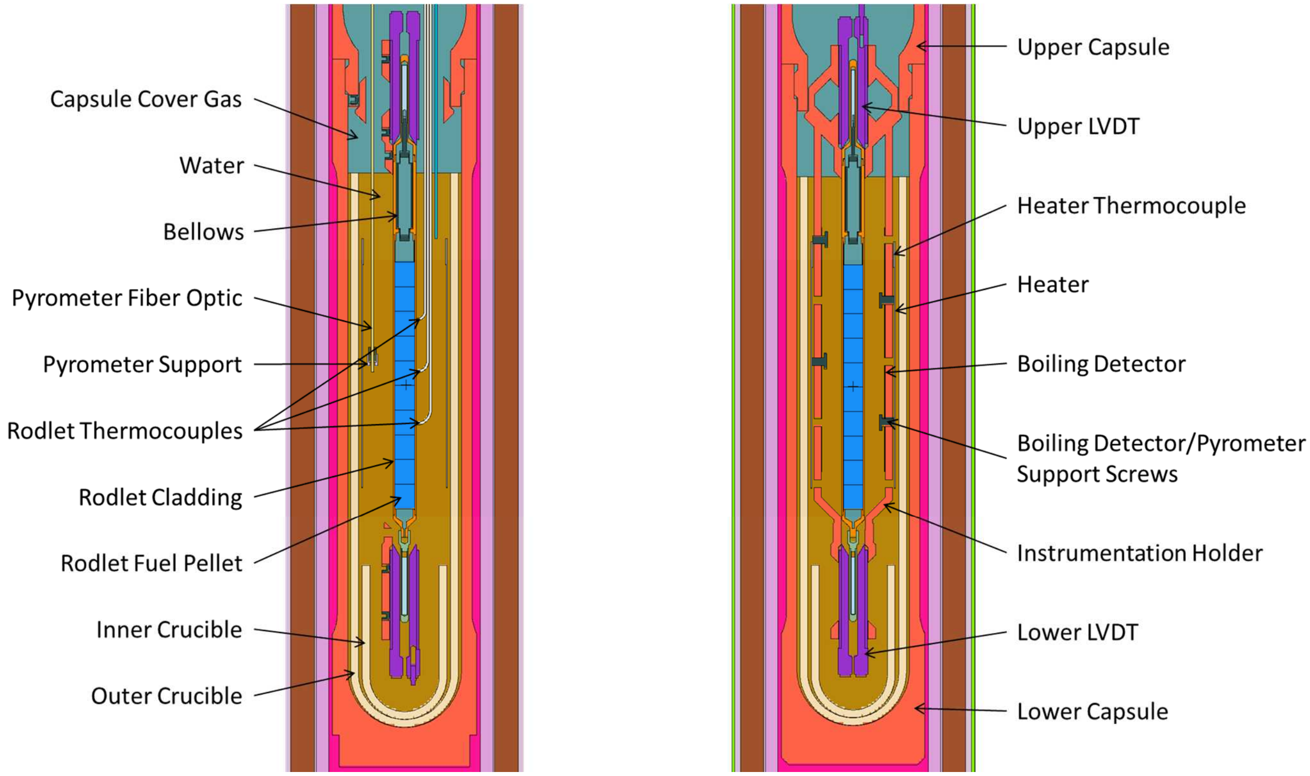


Figure 11. Close up of vertical cross section of lower MARCH-SERTTA capsule (Left South-North and Right West-East).

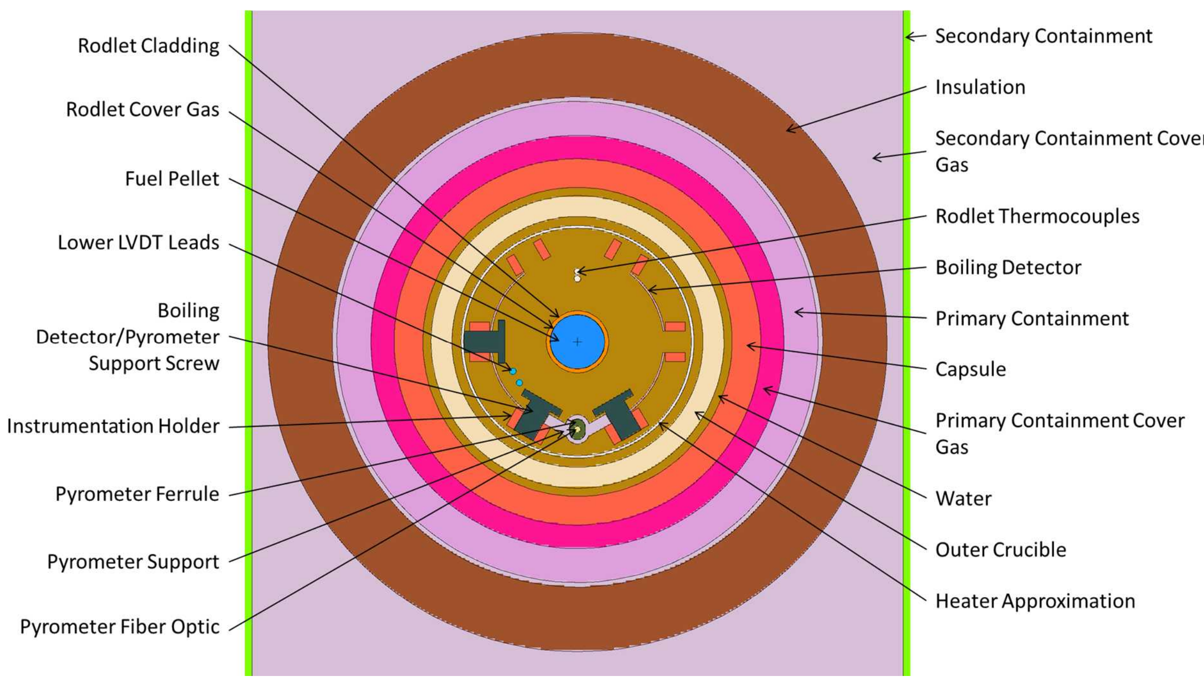


Figure 12. Horizontal cross section of MARCH-SERTTA capsule.

The MARCH-SERTTA design was analyzed using a variety of different configurations to determine an optimally achievable configuration. Using a natural uranium enrichment for the sample pellets, several models were created moving the vertical position of the capsule within the experiment section of the TREAT core while in a critical, steady state control rod configuration. The analysis showed that placing the top of the pellet stack at the core center plane would maximize the pellet PCFs and be low enough in the core to minimize interference of the control rod absorber sections on the flux. Enrichments of depleted

(0.30 wt.%), natural (0.711 wt.%), 3.2 wt.% , and 4.9 wt.% U-235 were then analyzed along with a 5 wt.% U-235 enriched, 70 GWd/MTU burnup fuel to determine the best option for the design. These enrichments were examined at varying experiment temperatures (the core temperature was held steady) in both water and steam environments to see the effect on HGRs and PCFs. Results can be seen in Table 1

Table 1. Average pellet PCF in W/g-MW at differing uranium enrichments and experiment temperatures in water and steam with the reactor core held at 20°C.

Pellet Enrichment	**20°C Water**	**200°C Water**	**280°C Water**	**320°C Water**	**140°C Steam**	**280°C Steam**
Depleted	0.326	0.254	0.226	0.209	0.160	0.163
Natural	0.667	0.511	0.451	0.420	0.311	0.314
3.2 wt.%	2.14	1.73	1.56	1.45	1.10	1.10
4.9 wt.%	2.80	2.33	2.12	1.99	1.54	1.55
High Burnup	1.37	1.22	1.15	1.10	0.784	0.797

The water in the capsule provides a significant increase in PCF for the pellets over PCFs observed in the MARCH-SETH experiment design [7]. In the case of MARCH_SETH the average pellet PCF for a 1.8 %Δk/k reactivity insertion, in helium, using 4.9 wt.% U-235 pellets, with a core temperature of 249°C was 1.726 W/g-MW. The increase seen in MARCH-SERTTA is likely caused by increased thermalization of fast and epithermal neutrons and increased neutron reflection around the test rodlet. As MARCH-SERTTA is intended for RIA testing, a minimum pulse width in TREAT is desired, but with the increased PCFs in the rodlet pellets from the water, it would be possible to have a PCF that would result in too much energy being deposited within the rodlet at certain enrichment levels.

Based on the results of the enrichment and temperature analysis and a need to meet RIA testing requirements, natural uranium (or close to natural ^{235}U/U enrichment) is likely the best candidate for MARCH-SERTTA fuel pellets. This enrichment level appears to be well aligned with the need for a minimum pulse width of ~90 ms, without depositing too much energy within the sample as higher enrichments may. Using natural uranium would result in approximately 667 J/g of energy deposited for a 1000 MJ reactor energy release during a clipped transient, near the 711 J/g regulatory limit, but would still allow for fuel melting during an unclipped transient. Following this determination, additional analysis was performed to examine the effect of increasing core temperature on the PCFs, with the results shown in Table 2 As can be seen, a maximum modeled PCF was observed with a room temperature experiment and a hot reactor core, with a trend of increasing PCF with increasing core temperature.

Table 2. Average pellet PCF in W/g-MW at differing reactor core and experiment temperatures in water with natural uranium pellets

Core Temperature(below)	**20°C**	**200°C**	**320°C**
20°C	0.667	0.511	0.420
200°C	X	0.578	X
320°C	0.806	X	0.515

X indicates configuration was not analyzed.

With the pellet enrichment and capsule position determined, a neutronic equivalent dummy (NED) was ascertained. The NED will be used in core characterization and physics measurements and needs to be neutronically equivalent to a fueled experiment. Because MARCH-SERTTA is rather complex, a simplified NED was desired to negate the need to have things like instrumentation, heaters, and water present to provide neutronic equivalence. It is also desired that the NED be low activation and durable to provide repeated use. For this, a grade 5 titanium slug was modeled in the shape of the outer crucible, but lengthened to fill the entire lower capsule cavity. The complete NED consists of the capsule and titanium slug, without any other instrumentation or internal parts and can be seen in Figure 13. The NED in this configuration provided a 0.010 %Δk/k reactivity difference from the natural uranium pellet configuration meeting the maximum +/- 0.05 %Δk/k change for neutronic equivalence.

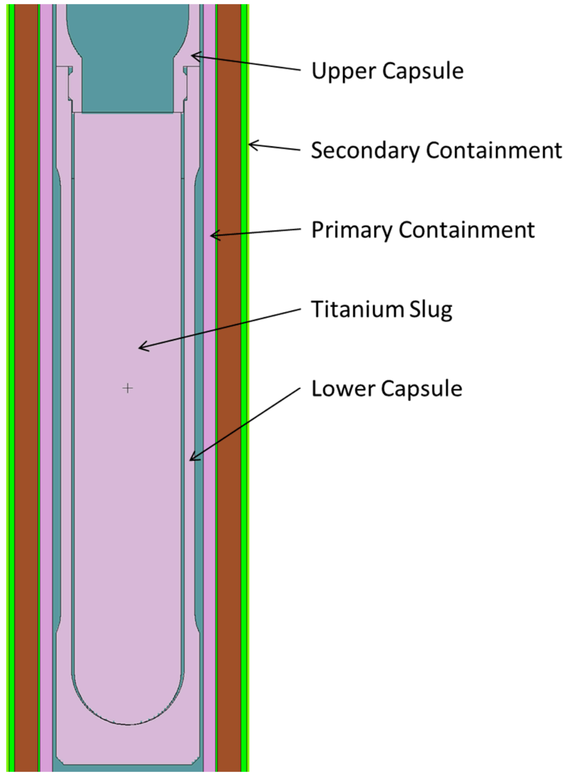


Figure 13. Vertical cross section (South-North) of MARCH-SERTTA capsule in NED configuration.

4. THERMAL HYDRAULIC CALCULATIONS

Preliminary Relap5-3D thermal-hydraulic calculations have been performed on the MARCH-SERTTA capsule to ensure safe operations during nominal and beyond-nominal conditions. A schematic of the MARCH-SERTTA device and a corresponding Relap5-3D nodalization is shown in Figure 14. The experiments will begin with an initial temperature of 200°C to accommodate polymer O-rings. Keeping the degree of subcooling consistent with PWR conditions results in an initial pressure in the device of 3.65 MPa (530 psi). The influence of the sub-PWR starting temperature and pressure on fuel performance figures of merit will be studied in the future, but the influence of starting at PWR conditions will be briefly discussed here.

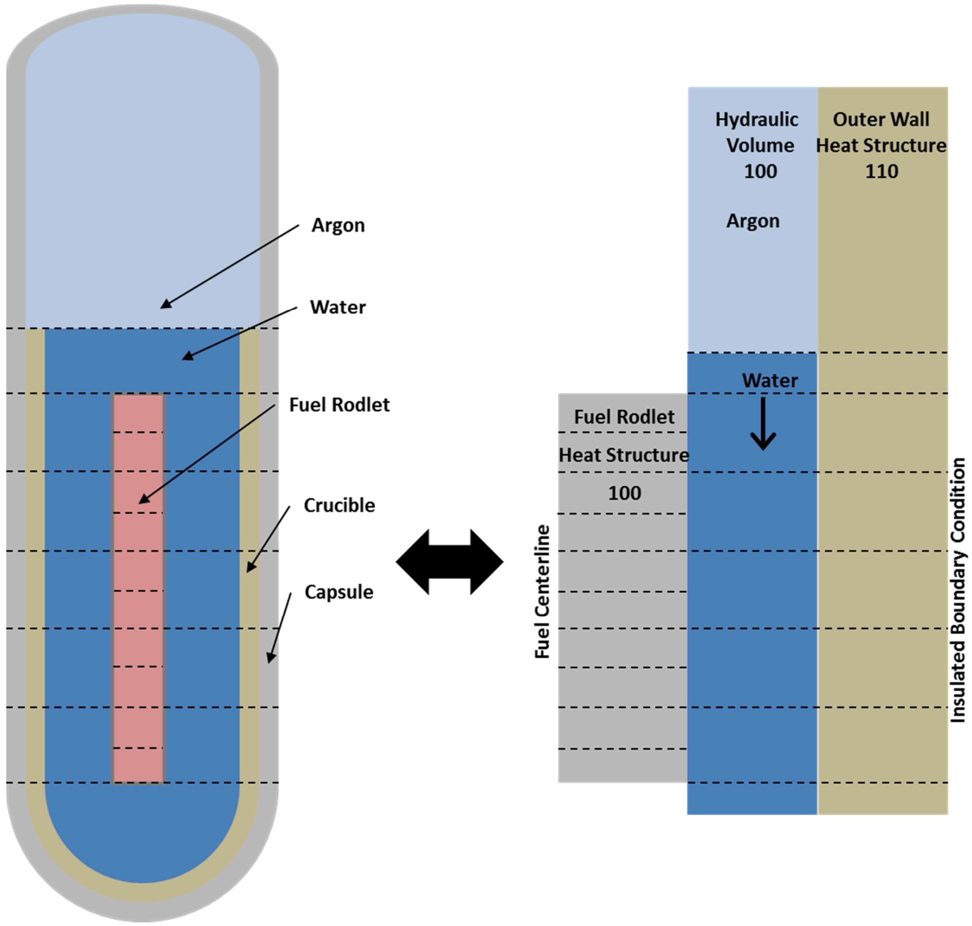


Figure 14. MARCH-SERTTA schematic and corresponding Relap5-3D model nodalization.

A planned nominal transient will constitute a 4.2% Δk/k clipped pulse resulting in 1380 MJ of reactor energy. With the corresponding energy coupling factor for natural uranium in 200°C water this pulse will deposit 711 J/g of energy into the fuel. This will results in 36.3 kJ of energy deposited into the fuel causing the peak vessel pressure to reach 4.72 MPa (684 psi) and a peak fluid temperature of 258°C. If the initial temperature and pressure were at PWR conditions (300°C and 15.5 MPa) the peak pressure is 16.76 MPa (2430 psi) and peak fluid temperature is 350°C.

The safety case with the MURA pulse (4.5% Δk/k unclipped) will result in 2925 MJ of reactor energy and 1527 J/g of energy deposited into the fuel. With the initial temperature and pressure of 200°C and 3.65 MPa the peak pressure and fluid temperature predicted by Relap5-3D is 7.48 MPa (1085 psi) and 287°C. The energy in this case is enough to melt the fuel resulting in some degree of fuel coolant interaction (FCI) that is not accounted for in this case. That will need to be studied further. If initial PWR conditions were considered the peak pressure is just under 21 MPa (3000 psi) and the peak fluid temperatures are 367°C.

During the safety case the temperatures of the water and cladding will be elevated to the point where excessive oxidation may occur releasing a significant amount of energy into the system. If all of the Zircaloy material in contact with the fuel completely oxidizes ~75 kJ of energy will be released which is almost equivalent to the nuclear energy produced during this case (~77 kJ). With this energy added to the water volumes in the Relap5-3D model the resulting peak pressure and fluid temperature is 13.6 MPa (1975 psi) and 346°C when starting from 200°C and 3.65 MPa, again no dynamic effects of FCI is taken into account.

The current design of the system includes the specimen holder (Figure 8) made from titanium which can undergo a similar exothermic reactor with water at elevated temperatures as zirconium. The amount of titanium present can result in an order of magnitude more energy than the zirconium reaction if all of the titanium reacts. Under these worst-case conditions there will be enough energy in the system to exceed super-critical water conditions.

The MARCH-SERTTA device has a built-in burst disc in the lid to relieve pressurize into the BUSTER pipe in the event of a significant pressure rise. The burst disc will be rated between 3-4 ksi. During normal test conditions ~711 J/g of energy deposited into the fuel the pressures remain well below the burst disc ratings for both 200°C and 300°C starting conditions to ensure no contamination of BUSTER during planned transients. Under the MURA conditions the pressures will still maintain below the burst pressure when starting at 200°C and will be right on the limit of 3 ksi when starting at 300°C. When accounting for the zirconium oxidation energy during the MURA transient the peak pressure can still maintain below the burst rating when starting at 200°C. At a staring temperature of 300°C the pressures will exceed the burst disc rating and vent into the BUSTER pipe. If significant titanium oxidation occurs there will also be enough energy to exceed the burst disc rating even when starting at 200°C.

5. STRUCTURAL CALCULATIONS

Preliminary calculations of the MARCH-SERTTA module represented the geometry using a finite element (FE) model with two main parts: the bottom SETH module and the top cap (SERTTA-specific). A 1262 psig (8.70 MPa) internal pressure of was applied to the module. This resulted in a maximum calculated stress of 9.9 ksi (68.3 MPa), which occurred in the thinnest wall section just above the threads in the threaded connection (see Figure 15). This maximum stress is only 14% of the yield strength of the Titanium Grade 5 material at 800°F (426.7°C) – showing significant margin against material yielding. This evaluation will be updated as the actual module temperature and pressure histories are calculated via thermal analysis.

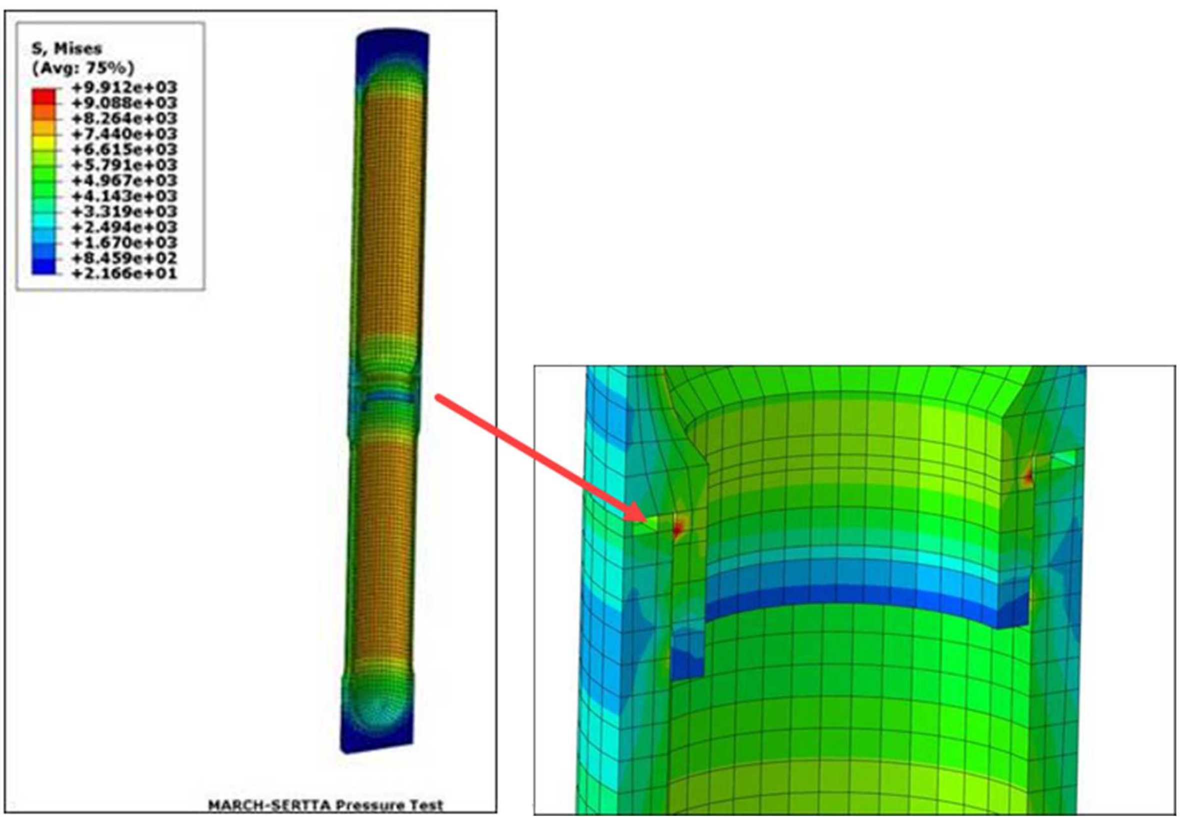
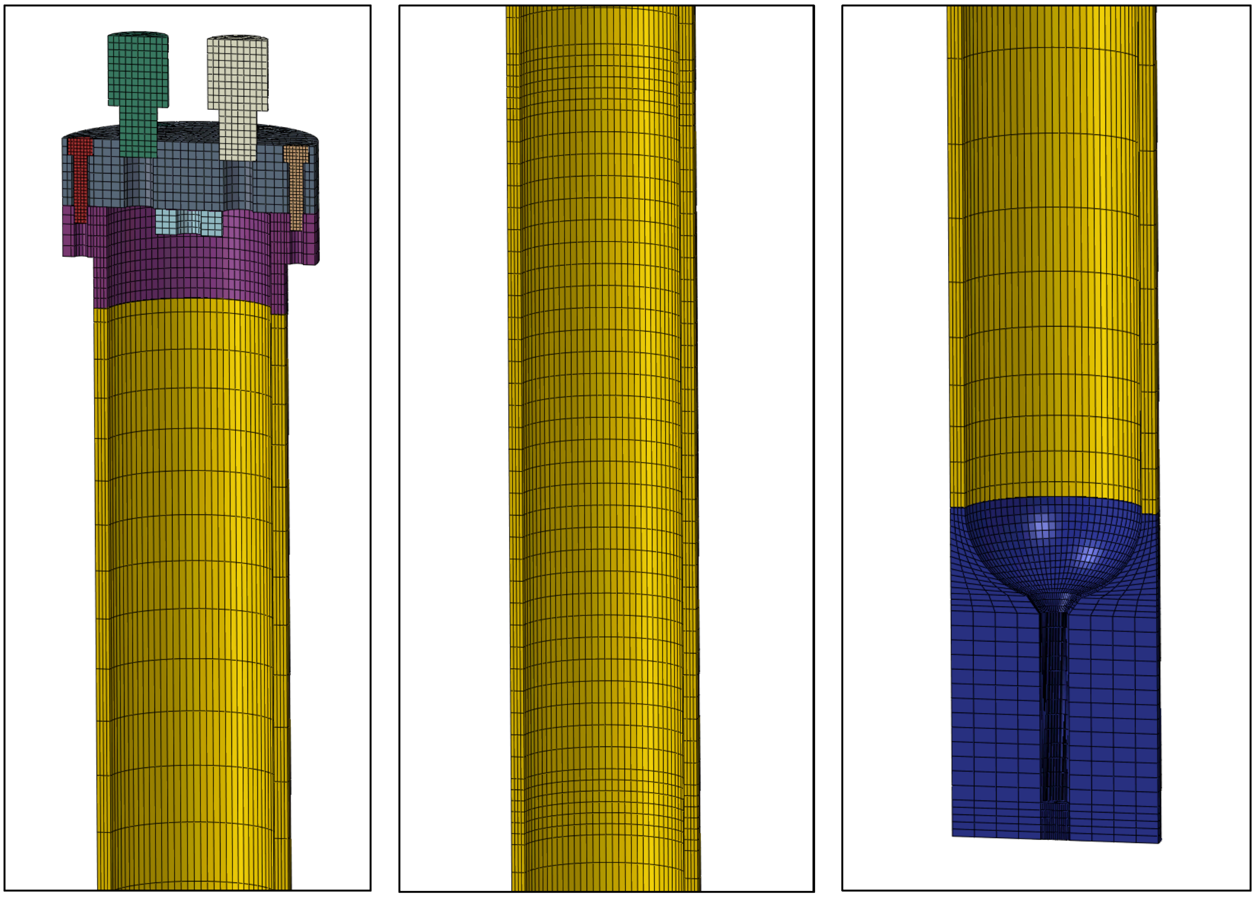


Figure 15. MARCH-SERTTA FE model results for a 1262 psig (8.70 MPa) internal pressure loading.

The MARCH-SERTTA module will be housed within the BUSTER containment structure. This structure has been modeled and evaluated previously for the SETH experiments (Ref. ECAR-3997 Rev. 1). That evaluation showed the BUSTER allowable internal pressure was limited by the closure screws (e.g., 277 psig [1.91 MPa] at 800°F [426.7°C]). Anticipating internal pressures that may exceed the BUSTER/SETH configuration, the BUSTER containment allowable internal pressure (i.e., pressure rating for a specified temperature) will be enhanced by using Inconel-718 closure screws. Figure 16 shows the FE model of the BUSTER containment used in the SETH experiment evaluation.



Top section | Middle section (adjacent to SERTTA module) | Bottom Section

Figure 16. BUSTER containment FE model.

The model of the BUSTER containment will be evaluated for pressure and thermal stresses as the actual loading history is calculated via thermal analysis.

6. CONCLUSIONS AND FUTURE WORK

The MARCH-SERTTA concept has been designed to a fair level of maturity and undergone analytic treatment to determine whether it can adequately support project objectives and satisfy its engineering requirements. The work described herein has concluded that the concept is valid and more detailed design is warranted including comprehensive analytic treatment of all typical nuclear experiment design considerations, physical construction of a prototype, and earnest project planning for conducting the first water-bearing experiments before the close of fiscal year 2019.

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